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Abstract

SODIUM-COOLED THERMAL REACTORS

Chauncey Starr (Atomics International, Canoga Park, California)
Sidney Siegel (Atomics International, Canoga Park, California)

The objectives of the development program for sodium-cooled thermal reactors are 1) attainment of low-cost electric power; 2) a reactor type having exceptional safety characteristics; 3) development of large sodium systems for future fast breeders. The U. S. Atomic Energy Commission program includes the operation and power expansion of the 20 MW(th)/6MW(e) Sodium Reactor Experiment since 1957, the 256 MWth/81 MW(e) Hallam Nuclear Power Facility on the system of Consumers Public Power District of Nebraska, and the development of major components, uranium monocarbide fuel and plant designs for 200 and 1000 MW(e) Sodium Graphite Reactor power stations.

Current designs utilize enriched UC fuel and produce steam at 170 atmospheres and 570C. Because of the low reactor pressure and the chemical affinity of sodium for iodine, there is little fission-product release from the system in event of fuel failure, thereby permitting conventional buildings. The high thermal efficiency, near 43%, results in good long-term fissile fuel utilization. A 500-MW(e) plant of this type can produce power, under USA private financing conditions, at a projected 5.7 mills/kWh.

The SRE began operation in 1957. The first core was unalloyed uranium metal, and operated at sodium temperatures up to 1000F(538C). In 1959, twelve fuel elements failed due to coolant flow blockage by in-leakage and decomposition of auxiliary hydrocarbon coolant. The damaged core elements were removed with remote-handling equipment, and operations resumed in late 1960, with a second core of metal Th/U alloy and the original Na coolant. Subsequent operations have provided extensive information on contaminant removal from Na systems, dynamic behavior of various fuel elements, and on performance of components in sodium.

The Hallam Nuclear Power Facility began power operation in July 1963. The first core is U/10% Mo alloy; the second fuel loading being fabricated is UC. With Na at 930F (500C) steam is produced at 60 atmospheres, 815F (435C). During preoperational tests, a tube failure occurred in an intermediate heat exchanger due to flow-induced vibration. All six exchangers were satisfactorily modified with internal bracing. The Zircaloy control-rod thimbles had to be replaced with stainless steel due to internal hydriding. All other major components and systems have performed satisfactorily.

The plant has operated up to 81 MW(e) under load-following conditions on a routine basis since October 1963. By January 1, 1964, about 80 million kWh of electric energy had been produced. The plant availability during the previous month of December 1963 was 99.4%.

Large SGR designs are now being developed in the 200 to 1000 MW(e) range. The fuel is stainless steel clad UC operating at a power density of about 50 kW/kg. With sodium coolant temperature at 1500F (620C), steam pressures can be between 170 atmospheres and supercritical. The graphite moderator for these plants is now contained in a single large stainless steel calandria vessel. The heat-transfer systems are designed for about 500 MW(th) each, using single large 2200 liter/sec pumps, and modular once-through steam generators. A 200-MW(e) plant uses one such loop, a 1000-MW(e) plant five similar loops. Advanced variants of the SGR, e.g., coupled thermal-fast cores, may provide near-term intermediate steps to fast reactors.